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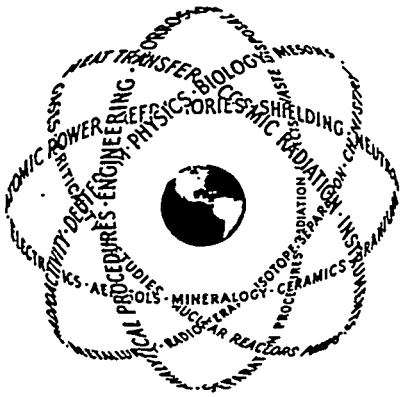

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RADIOACTIVE WASTE PROCESSING AND DISPOSAL (EXCERPT)

A Bibliography of Selected Report Literature Available to the Civilian Application Program

Compiled by
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9
PUREX WASTES TREATMENT CHEMICAL FLOWSHEET. H. King, V. J. Reilly, and J. R. Flanary. Jan. 31, 1952. Changed from SECRET Jan. 4, 1956. 12p. (CF-52-1-195). CONFIDENTIAL
A flowsheet for treatment and disposal of radioactive wastes (except head end and tail end) associated with processing uranium (450 Mwd/T) per Purex Process Flowsheet #3 is presented.

10
A DISPOSAL SCHEME FOR DAREX PROCESS WASTES. M. L. Hyman. Apr. 25, 1957. 4p. (CF-57-4-132). CONFIDENTIAL
The disposal scheme presented proposes to evaporate and calcine the wastes to the metal oxides with the simultaneous recovery of the nitrate values for recycle in the plant. The oxides would then be reduced (with H_2, C, or other reducing agent, or by a thermite reaction) to the metals. The final form of the material would be a fine powder of "stainless steel" containing the fission products, or, if the thermal environment is suitable a melt of solid "stainless steel" containing fission products could be produced. No flowsheets, diagrams, etc., of the scheme are given.

11
LIQUID WASTE DISPOSAL AT OAK RIDGE NATIONAL LABORATORY. F. N. Browder. March 28, 1949. Changed from SECRET Jan. 12, 1956. 49p. (ORNL-328). CONFIDENTIAL
The four types of liquid waste (radiochemical, metal waste, warm waste, and process waste) from ORNL processes are described, and the method of handling each is explained. The waste tanks are listed and the status of the storage facilities described. Recommendations for correcting existing faults in the waste disposal system are made, and possible future developments in waste decontamination are listed. 56 references.

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62
REPORT OF TECHNICAL DIVISION SECTION I FOR MONTH ENDING OCTOBER 20, 1948. F. L. Steahly. Oct. 20, 1948. Changed from SECRET Dec. 29, 1955. 30p. (CF-48-11-224) CONFIDENTIAL
Progress is reported on Redox studies; second U recovery cycle in laboratory and semiworks runs; Pu oxidation and distribution; waste treatment by ion exchange; decontamination of 25 and plant radiochemical wastes; Th recovery by solvent extraction; theoretical extraction studies; equipment testing and development; bibliography of chemical process development; section 23 and 25 reports; Redox second-cycle U flowsheet; flowsheet for Al decontamination from 25 wastes by ion exchange.

63
AN EVALUATION OF MATERIALS FOR LINING EARTHEN BASINS FOR CONTAINING RADIOACTIVE SOLUTIONS. G. A. West, F. L. Rodgers. p. 27-8 of UNIT OPERATIONS STATUS REPORT FOR MARCH 1955. (Unit Operations Sect., Chemical Technology Div.). W. K. Eister, ed. 60p. (CF-55-3-190) CONFIDENTIAL.
Permeometer tests to determine the leakage of simulated waste through asphalt, soil, ball clay, and shale are summarized. The tests indicate that the leakage through a 1/2" asphalt layer is negligible. Laboratory tests of 1/16" asphalt and permeometer tests of 1/2" thick asphalt show that tap water at 77°C has a progressive deteriorating effect. Water penetrated the 1/16" asphalt in 30 days and the top layer of the 1/2" asphalt deteriorated. This suggests that leakage may occur after long time exposures.

64
AN EVALUATION OF MATERIALS FOR LINING EARTHEN BASINS FOR CONTAINING RADIOACTIVE WASTE. G. A. West. p. 66-7 of UNIT OPERATIONS SECTION, CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR JUNE 1955. W. K. Eister. 94p. (CF-55-6-180) CONFIDENTIAL
Results of tests on four tars, Barrett Roofing Pitch, pipeline enamel, millwrap enamel, and waterworks enamel show that they do not have the resistance of asphalts against a simulated waste (1.6M $Al(NO_3)_3$, 8.2 N acid) and against tap water at room temperatures.

65
RADIOCHEMICAL WASTE DISPOSAL. p. 78-86 OF TECHNICAL DIVISION REPORT FOR QUARTER ENDING SEPTEMBER 1, 1948. Stuart McLain. Sept. 1, 1948. Changed from SECRET Jan. 24, 1956. 118p. (ORNL-140) CONFIDENTIAL.
Preliminary work is reported on a program to reduce the amounts of activities in the plant wastes. Design studies are reported for an evaporator to concentrate the most active non-fission metal liquid wastes to increase storage capacity. A survey of the best methods of eliminating or removing particles of UO_2 from the air discharged from the pile is presented.

66
RADIOACTIVE GAS SEPARATION. p. 55-8 OF PUREX PILOT PLANT QUARTERLY REPORT FOR MAY, JUNE, AND JULY, 1951; CHEMICAL TECHNOLOGY DIVISION. D. O. Darby. Nov. 14, 1951. 74p. (ORNL-1115(Del.)) CONFIDENTIAL
Equipment for the radioactive gas separation process is described and a schematic diagram of the system is shown.

67
WASTE TREATMENT STUDIES. p. 13 of CHEMICAL TECHNOLOGY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1954. June 29, 1954. 26p. (ORNL-1708(Del.)). SECRET
It is now planned to store the radioactive wastes from Hope type processes and the ORNL 25 recovery process in open basins in the earth rather than in the carbon steel, stainless steel, or concrete tanks now used for similar wastes. All the commercially available asphalts and tars that have been considered as lining material for these basins have deteriorated when they were exposed to activities considerably below those anticipated in the waste solutions. Two special modified asphalts with softening points of 278 and 256°F showed a 30% increase in volume when γ irradiated to 5×10^8 r (equivalent to 20 to 100 yr of exposure, through 2 ft of earth shielding, to 500-c/gal waste) and were severely attacked when immersed in boiling synthetic Hope type process waste (1.6M $Al(NO_3)_3$, 0.2M HNO_3, 0.02M H_2SO_4) for 3 hr. Four modified tars with softening points of 237, 199, 192, and 158°F were not affected by exposure to 5×10^8 r of γ radiation or by immersion for two days in synthetic Hope process waste solution at 150°F. (The applicable portion of this report appears in its entirety.)

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REFERENCES

68
CERAMIC ASPECTS OF WASTE DISPOSAL. p. 122-127 of METALLURGY DIVISION SEMIANNUAL REPORT FOR THE PERIOD ENDING APRIL 10, 1955. Nov. 7, 1955. 144p. (ORNL-1911(Del.)) SECRET

Experiments to investigate the possibility of incorporating radioactive isotopes contained in Hope solution into a ceramic body are reported. The method studied involves mixing the liquid waste with clay, limestone, soda ash, or other material to form a gel or slurry which is subsequently dried and sintered to produce a mass in which the isotope may be fixed and from which the isotope cannot be leached. No results of the experiments are given. Experiments to test the ability of concrete to contain Hope solution are also reported in progress. Results of experiments to indicate the power generation required to bring a clay-flux mixture containing fission product waste to 900°C are shown in graphical form. The time-temperature plot in the results indicate that ~100W would have brought the temperature up to 900°C in ~29 days. Procedures, computations, and results of viscosity studies of selected clays for use in lining waste disposal pits show that the Hope solution increases the viscosity of Panther Creek bentonite, whereas W-8 solution decreases it. Both solutions lower the viscosity of the Wyoming bentonite very considerably, the W-8 even more effectively than the Hope. The viscosity of the Gleason ball clay is increased quite appreciably by both solutions. Mixtures of 10, 25, and 30 g of Gleason ball clay in 100 cc of distilled water show a consistent increase in apparent viscosity with increasing clay content.

69
WASTE DECONTAMINATION. p. 11-12 of CHEMICAL TECHNOLOGY DIVISION SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1956. June 22, 1956. 57p. (ORNL-2079) CONFIDENTIAL

The present philosophy on waste treatment for the Hope project is to remove the long-lived radioactive fission products Cs and Sr and to store the remaining waste in inexpensive pits. A schematic diagram of the diban-ion-exchange process for removing long-lived fission products from Al-containing waste is shown in which synthetic waste was decontaminated by a factor of 220. In studies on removal of $Al(NO_3)_3$ from wastes containing fission products by crystallization, tracer Cs and Sr were separated from Al by factors of greater than 60 and 130, respectively, in one crystallization. In a second crystallization the DF, S were greater than 60 and 10, respectively. A schematic diagram for the crystallization method is included.

70
WASTE TREATMENT AND DISPOSAL. R. J. McNamee, F. L. Rogers, S. G. Kent, I. R. Higgins. p. 40-2 of CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR MARCH 1957. Aug. 12, 1957. 96p. (ORNL-2307) CONFIDENTIAL.

A proposed new method for permanent radioactive waste disposal is briefly reported. The method consists of mixing solids resulting from evaporating waste solutions with asphalt, thereby forming a mass from which it is hoped activity cannot be leached. A mixed (anion- and cation-exchange) resin bed has been proposed for treating the water of water-cooled reactors for protection against corrosion. A schematic diagram of the unit is included. The adequacy of the resin method is discussed.

71
WASTE TREATMENT STUDIES. R. J. McNamee, A. R. Irvine. p. 48 of CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR APRIL, 1957. Aug. 12, 1957. 72p. (ORNL-2324) CONFIDENTIAL

Studies on disposal of solid wastes by mixing with asphalt, as previously presented in (ORNL-2307), are reported. Continued tests on leaching of fission products from the asphalt mixture revealed that after an initial build up of activity in the leach water equal to 0.2 and 0.4% of the β and γ activity, respectively, the activity leveled off at an ~ constant value, indicating that little further leaching is taking place. The results of tests for 4 consecutive weeks are given in tabulated form.

72
WASTE STUDIES. I. R. Higgins. p. 32-3 of CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR JUNE 1957. Nov. 6, 1957. 75p. (ORNL-2362). CONFIDENTIAL.

Results of waste treatment testing of high level solvent extraction raffinate by electrolytic destruction of nitrate, caustic regeneration, and plating of the Ru using an Fe cathode and stainless steel anode showed nitrate destruction of 60-80%. Ru was plated on the cathode. ORNL wastes run about 20g nitrate/liter, so costs would be ~1.5¢/gal for electricity (at 0.5¢ kwh).

73
WASTE STUDIES. p. 30-1 of CHEMICAL TECHNOLOGY DIVISION MONTHLY PROGRESS REPORT FOR AUGUST 1957. Jan. 9, 1958. 83p. (ORNL-2400). CONFIDENTIAL

A new process and flowsheet for ORNL waste is presented. At the present time these wastes are stored in ground surface pits. Naturally occurring Conasauga shale possesses sufficient ion exchange capacity to sorb all the fission products except Ru, but the Ru, together with the nitrate ions, which escapes directly to the ground water and may eventually contaminate the environment. In the new process, waste at a rate of about 5,000 gal/day would be sent to either a continuous ion exchange contactor or a shale-filled pit where Cs, Sr, and the rare earths would be removed. Data show that a 6 in. dia. column would be required. Based on recent determinations by the Health Physics Division, approximately 35 yd^3 of shale per year would be required as an alternative. For the removal of nitrate and Ru, two types of electrolytic cells are being considered. An acid-base cell using anion- and cation-exchange membranes would require about 35 ft^2 each of electrode area and a 60-kw power supply. A nitrate reduction cell would be simpler in construction, not requiring membranes and compartments, but possessing a total of about 280 ft^2 of electrode area and a 500-kw power supply. The latter cell may be somewhat more efficient for Ru plating. On the other hand, if Ru removal only is desired without nitrate or caustic destruction or recovery, a much simplified electrolytic cell may be used. Considerably more laboratory work will be required before the final flowsheet conditions and equipment may be chosen. (The applicable portion of this report appears in its entirety in this abstract).

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113
EVALUATION OF FULL SCALE SAVANNAH RIVER PROJECT EVAPORATOR. W. B. Watkins. Nov. 19, 1951. Changed from SECRET Nov. 29, 1955. 20p. (CF-51-11-113). CONFIDENTIAL

Data obtained from the operation of the full-scale Savannah River type evaporator now installed in the ORNL pilot plant are summarized. The program was initiated to obtain data that would guide current and future evaporator design in connection with the Savannah River Project. The principal variable considered was the effect of boil-up rate on the over-all decontamination factor obtainable from the evaporator plot to the condensate.

114
RESULTS OF EVALUATION OF THE SAVANNAH RIVER TYPE EVAPORATOR USING SYNTHETIC HIGH ACTIVITY WASTE. W. B. Watkins. Dec. 14, 1951. Changed from SECRET Nov. 18, 1955. 6p. (CF-51-12-118). CONFIDENTIAL

The results of this study show that the Savannah River high activity evaporator can be expected to give an over-all decontamination factor (evaporator liquid to condensate) of at least 3.5×10^5 when operated at its designed capacity of 87.8 lb/hr/ft^2 boil up rate and 3.1 ft/sec superficial velocity in the column. At boil up rates of 115 to 125 lb/hr/ft^2 and column superficial vapor velocities of 5 to 7 ft/sec, (which are 150 to 200% of designed capacity) the over-all decontamination factor is likely to be from 5×10^3 to 1×10^4. While it was impossible to operate the ORNL evaporator at a boil up rate of 187 lb/hr/ft^2 which would simulate the low activity unit, it is believed that the over-all decontamination which could be expected from this unit would be no greater than the value of 1×10^4 which was obtained at a 115 to 125 lb/hr/ft^2 boil up rate. A heavy brown precipitate, believed to be MnO_2, which formed upon initial heating of the synthetic high activity feed solution, caused considerable difficulty in sampling and in cleaning the evaporator. Traces of the solids were found in the condensate samples obtained from tests at boil up rates over 115 lb/hr/ft^2 and may have contributed to a lowering of over-all decontamination factor since brief experiments showed that the precipitate carried from 50 to 90% of the activity in the feed solution.

115
DISTILLATION OF PUREX WASTES. R. G. Mansfield. April 24, 1952. Changed from SECRET Nov. 10, 1955. 18p. (ORNL-1472). CONFIDENTIAL

Experiments on distillation of Purex waste containing fission products showed that the radioactivity in the distillate is due principally to volatilized Ru. This can be minimized by keeping the still-pot HNO_3 concentration below 9M.

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145
FEASIBILITY REPORT ON RECOVERY OF FISSION Zr AND Cb FROM REDOX PROCESS WASTE SOLUTIONS USING SILICA GEL. W. H. Baldwin, J. H. Gross, C. E. Higgins, H. W. Kohn, J. M. Ruth, A. W. Smith, and R. E. Wacker. Sept. 28, 1949. 35p. (CF-49-9-178). SECRET

A series of runs with a Cm^2 cross-section silica gel column has shown that a process for the separation of Zr-Nb fission activity from 1AW Redox waste solution is feasible. The feed used in these test runs was the 1AW waste from the Redox Pilot Plant Operations at ORNL. A proposed flowsheet for operation with Hanford Redox waste is given. The indicated overall yield is 80 to 85%. This should amount to about 0.40, 0.31, and 0.26 megacuries of activity after 10, 25, and 50 days cooling, respectively. The largest proportion of the indicated 15 to 20% loss occurs in the adsorption step, and appears to be due to a slow attainment of equilibrium at the 30°C temperature used in the column test runs. Batch studies have indicated that equilibrium is reached much faster at higher temperatures; so it appears that the yield may be raised to around 95% by operating the adsorption step at higher temperatures or by operating at slower flow rates per unit cross-sectional area. Six successive runs were made with one column with no indication of decreased silica gel adsorption efficiency.

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151
IDAHO CHEMICAL PROCESSING PLANT LIQUID WASTE FACILITIES: DESIGN REPORT. F. N. Browder. Aug. 20, 1954. Changed from SECRET June 14, 1955. 272p. (ORNL-1687). CONFIDENTIAL

The liquid waste facilities of the Idaho Chemical Processing Plant are described in detail, the basis of the design is outlined, and early performance data from operation of the completed plant are presented. Only the facilities for liquid waste handling, exclusive of sanitary waste, are covered.

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166
COMPARISON OF THE THOREX AND REDOX AQUEOUS WASTE SYSTEMS. A. T. Gresky and R. P. Wischow. June 17, 1954. Changed from SECRET Feb. 15, 1957. 6p. (CF-54-7-254). CONFIDENTIAL

The results of the investigation indicated that the Thorex hot waste volumes should be at least a factor of 5 lower than Redox wastes, use of stainless steel in lieu of mild steel tanks for storage of the highly acid-deficient AP concentrate, and anticipated Al wastes from a second Th cycle, would decrease the cost savings factor from about 5 to 2.5, and storage of the concentrated Thorex AP stream may become mandatory if the Pa^{233} is to be recovered as U^{233} after cooling; i.e., in the eventuality that short-cooled Th is to be processed without the inclusion of a Pa recovery step.

167
HEAT TRANSFER IN WASTE BASINS. S. H. Jury. Aug. 11, 1955. Changed from SECRET Apr. 19, 1956. 17p. (CF-55-8-76). CONFIDENTIAL

Estimates are made of the heat transfer in proposed out-of-door radioactive waste basins. Calculations are outlined, and complete numerical results are included in appendices.

168
CALCULATIONS CONCERNING ALTERNATE WASTE DISPOSAL METHODS. p.143-149 of A CHEMICAL REPROCESSING PLANT FOR A NUCLEAR POWER ECONOMY. (PROJECT HOPE). R. A. Charpie, J. Halperin, R. J. Klotzbach, J. R. McWherter, F. Nelson, E. L. Nicholson, C. H. Odom, R. W. Stoughton, E. P. Wigner, and M. R. Zeitlin. Feb. 5, 1954. Decl. June 19, 1959. 166p. (ORNL-1638). UNCLASSIFIED

Calculations are summarized for the computing of the total amount of shadow shield required for personnel protection when the storage scheme is an open tank or pit. Preliminary estimates of alternate liquid waste storage tank installations have been prepared to obtain an order of magnitude of costs. The estimates cover only the purchase, erection, and earth or concrete shielding for a tank farm consisting of three 500,000 gal capacity tanks. The tanks considered are as follows: (1) spherical of mild steel construction internally braced to withstand 10 ft of earth covering, (2) cylindrical of mild steel construction internally braced to withstand 10 ft of earth covering, (3) cylindrical of mild steel construction built to withstand internal pressure of liquid only, and (4) cylindrical of stainless steel construction (347) to withstand internal pressure of liquid only. The total costs of the four systems as outlined above are as follows and are given in detail in tabulated form: underground spherical, $528,000; underground cy-

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174
SHIPPING COSTS OF SAVANNAH ENRICHED FUEL VIA COMMERCIAL CARRIER. R. J. Klotzbach. May 20, 1954. 10p. (CF-54-5-188(Del.)). SECRET

The cost of shipping Savannah River irradiated enriched fuel via commercial carrier to ORNL and to the ICPP has been estimated for both Railway Express and Motor Freight shipment. The total annual shipping costs for each type of carrier to both destinations at various process rates is shown.

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179
A STUDY OF THE CONTRIBUTION OF THE RALA PROCESS TO ATMOSPHERIC CONTAMINATION AT ORNL. R. L. Bradshaw and W. D. Cottrell. Nov. 1, 1954. 55p. (CF-54-11-186). SECRET

Data on the particulate contamination of the atmosphere at the Oak Ridge National Laboratory were correlated with laboratory processes and reactor operations for the period from March 1949 to June 1954. A significant correlation was found to exist between one of the chemical separation processes, RaLa, and peaks of particulate activity on the Laboratory area. The RaLa process was monitored during two complete cycles of operation. The contribution of the process to the general atmospheric contamination of the Laboratory area was determined. The activity was identified and its release to the atmosphere was investigated. It was concluded that the major portion of the contaminated RaLa effluent released to the stack does not fall out or diffuse in the immediate environs, and hence, the stack is not the primary means by which RaLa contributes to atmospheric contamination. A serious offender was found to be an underground liquid waste storage tank in the south tank farm. This tank is used to store RaLa wastes and is vented to the atmosphere. Atmospheric contamination in the immediate vicinity of this vent was observed to reach values as high as 2×10^{-5} μc/cc, β activity, averaged over a 24-hr period.

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200
DESIGN OF AN ADSORBER FOR REMOVAL OF KRYPTON AND XENON FROM DISSOLVER OFF GAS STREAM [FROM CPP 601]. J. M. Holmes. June 20, 1951. 29p. (CF-51-6-70). SECRET

The calculation for the design of an adsorber for the removal of Kr and Xe from the dissolver off gas stream is given. It appears that the three-month charcoal bed would be the most economical. It would be possible to use vacuum as insulation but the mechanical design problems and the maintenance problem in a hot area indicate that the use of insulation would be more economical and would simplify operation.

201
DESIGN OF THE DISSOLVER OFF-GAS SYSTEM FOR THE IDAHO CHEMICAL PROCESSING PLANT. John M. Holmes. Nov. 5, 1952. Changed from SECRET Apr. 4, 1956. 41p. (CF-52-11-39). CONFIDENTIAL

Pertinent calculations for the WN System adsorption units have been carried out on heat transfer, heat exchangers, and adsorption beds. Static adsorption and dynamic adsorption data are presented for Kr, Xe, Co, and N_2 on charcoal.

202
DESIGN OF A UNIT FOR THE REMOVAL OF IODINE FROM THE DISSOLVER OFF-GAS STREAM IN THE IDAHO CHEMICAL PROCESSING PLANT. William L. Carter. Nov. 5, 1952. 33p. (CF-52-11-48). SECRET

Iodine may be removed from the dissolver off-gas stream by passing the gas stream at an elevated temperature through a bed of ceramic porcelainized berl saddles which have been coated with $AgNO_3$. Any I_2 present will react quantitatively with the $AgNO_3$. The optimum operating temperature is 375°F. Design specifications and operating procedures for the I_2 removal unit are included.

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203
PILOT PLANT RADIOACTIVE GAS SEPARATION PROCESS EQUIPMENT AND OPERATING INSTRUCTIONS.
W. G. Watkins, W. L. Poe, and D. R. Lindow. Nov. 12, 1952. 45p. (CF-52-11-82). SECRET

The Radioactive Gas Separation Process (RAGS) to remove radioactive elements, particularly Kr^{85} from the gas evolved during the dissolution of irradiated fuel units, was installed in the ORNL pilot plant to confirm the feasibility of the laboratory flowsheet and to secure sufficient data to support an economic study of a plant scale installation. The process was operated in distinct phases; (1) the dissolver off-gas was continuously processed through the entrainment separator, condenser, I^{131} absorption tower, filter, acid tower and discharged into the gas holder; (2) the gas was continuously processed through the caustic scrub tower, Al_2O_3 dryer, cold trap and silica gel adsorption unit; (3) the Kr and Xe were desorbed from the silica gel bed and stored. The off-gases from 24 runs were processed through the equipment. The adsorption of Kr on silica gel

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